Status of the $^9\text{Be}$-EFF-3.0/NMOD=3 data processing with NJOY/ACER

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Recent evaluation by V. Pronyaev, S. Tagesen, H. Vonach (EFF-DOC-689)

- Calculation and adjustment of neutron emission channels as partial (n,2n) cross sections
- Including energy-angle distributions (and covariances)
- 16 channels for neutron emission, 17 for alpha emission
\(^9\text{Be EFF3.0/NMOD=3}\)

- ENDF-formatted data
  - Partial cross sections are given as MT875 to MT890
  - Energy-angle distributions are stored in subsections of MF6 for neutrons and alphas (LAW=7), multiplicity 2 is used explicitly
  - Included also the rather sharp neutron emission line of \((n,n''x)\) as a separate distribution in MT876
  - The now redundant \((n,2n)\)-cross section (MT16) is given as a full consistent sum of the individual channels
Objective

- Use of individual neutron emission channels in MC transport and sensitivity calculations
- ENDF-formatted data as given in MT875-890 (file 3 and 6) should be processing into a ACE-file for subsequent use in MCNP
- Present status of NJOY/RECONR and ACER does not allow to treat these data
Processing with ACER

- PENDF-output of RECONR added now by inclusion of MT875-890
- ACER dosimetry processing (only MT3 data) available by inclusion of MT875-890 in UNIONX
- ACER fast data processing (transport)
  - Allow for neutron emission and other particle emission in MT875-890
  - Increase ACE storage significantly
  - Allow for combining both neutron angular distributions in MT876 (option newfor=1 to use arbitrary cosine bins)
Processing with ACER

- **MT876: $^9\text{Be}(n,n''n2\alpha)$**
  - $2^{\text{nd}}$ exited Level ($2.43\text{ MeV}$), $\Gamma=770\text{eV}$
  - *Contributes between 50% at low energy and 20% at high energy to total neutron production cross section*
  - *Branches to ground state of $^8\text{Be}$ (7%) or three-body-break-up (93%)*
  - Angular distributions of inelastically scattered neutron and of second neutron are given independently
  - *ACE-format allows only single angular (but multiple energy) distributions*
Angular distribution of neutrons from MT876
\((E = 14 \text{ MeV})\)

- ○ inelastically scattered neutron
- ● second neutron
- ✱ all neutrons

\(\text{pdf [a.u.]}\)

\(\mu\)
Energy distribution of inelastically scattered neutron from MT876 (E = 14 MeV)
Energy distribution of second neutron from MT876
(E = 14 MeV)
Processing with ACER

- MT16 (n,2n)
  - is now redundant and has to be removed for transport calculations
  - could be easily achieved by elimination of all entries in the ENDF-file (includes MF1 directory, MF3 cross section, and MF6 distributions)

- Total size of ACE-file: 33 MB
  total size of ENDF-file: 17 MB
Checking the ACE-file

- Completeness
- Basic reaction parameters
- Cross sections (MF3) including total
- Angular distributions; MT876 after adding both neutron contributions
- Energy distributions
- Alpha-production: yields, contributing MTs, angular and energy distributions
- Todo: Application to MCNP-transport benchmarks
Reactions in ACE-Output

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